ACCESSION #: 9206250028
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Nine Mile Point Unit 2 PAGE: 1 OF 18

DOCKET NUMBER: 05000410

TITLE: Multiple Engineered Safety Feature Actuations and the Discovery of Design Deficiency as a Result of the Loss of Offsite Power due to Inadequate Work Practices
EVENT DATE: 03/23/92 LER #: 92-006-01 REPORT DATE: 06/15/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: 5 POWER LEVEL: 000

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(ii) and 50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On the morning of March 23, 1992, with the reactor at 0 percent power, and the reactor mode switch in the "REFUEL" position (Operational Condition 5) with the core partially offloaded, Nine Mile Point Unit 2 (NMP2) experienced a loss of Line #5 and a subsequent loss of Line #6, the plant's 115 KV offsite power sources. As a result, numerous Engineered Safety Feature Systems actuated, Control Room annunciation was lost, and a Reactor Protection System Logic full scram signal was received. The loss of Control Room annunciation mandated entry into an "ALERT" as specified by the Site Emergency Plan. Additionally, the High Pressure Core Spray (HPCS) Diesel Generator tripped following the sequential Loss of Offsite Power (LOOP).

The root causes for the loss of plant offsite power have been attributed to inadequate work practices and inadequate operator training.

The cause for HPCS Diesel Generator trip was the occurrence of an event that was not considered in the plant's design bases.

Corrective actions included: upgrading Operator and Maintenance personnel requalification training programs; revising operating procedures; completing Engineering evaluation of critical plant responses; issuing site direction on management expectations on Administrative Procedure use; and completing a modification to the High Pressure Core Spray Diesel Generator cooling water supply valve isolation logic.

END OF ABSTRACT

I. DESCRIPTION OF EVENTS

On the morning of March 23, 1992, with the reactor at 0 percent power, and the reactor mode switch in the "REFUEL" position (Operational Condition 5) with the core partially offloaded (Second Refuel Outage), Nine Mile Point Unit 2 (NMP2) experienced a loss of Line #5 and a subsequent loss of Line #6, the plant's 115 KV offsite power sources. As a result, numerous Engineered Safety Feature (ESF) Systems actuated: Divisions II and III Emergency Diesel Generators auto started; a Secondary Containment isolation signal initiated; Train B of the Standby Gas Treatment System initiated; and a full reactor scram signal was received. Additionally, Control Room annunciation was lost, mandating entry into an "ALERT" as specified by the Site Emergency Plan.

NOTE: On March 24, 1992 an NRC Augmented inspection Team arrived at the plant to review and determine the circumstances that led to this event, its causes, safety significance, and adequacy of Niagara Mohawk's response to the event. In reply, Niagara Mohawk has submitted an assessment of corrective actions taken and actions to be taken, relative to the significant occurrences bounded by this event but not included in the text of this Licensee Event Report. This report was submitted to the NRC on May 13, 1992.

NMP2 is supplied with two redundant 115 KV offsite power sources which are designated as Line #5 and Line #6. There are provisions provided to interconnect the plant electrical distribution system to either Line #5 or Line #6. in the plant's 115 KV switchyard, there are three stepdown transformers, namely Reserve Station Transformer (RSS) 2RTX-XSR1A, 2RTX-XSR1B, and Auxiliary Boiler Transformer 2ABS-X1. The two RSSs supply safety related loads and normal station loads during shutdown periods. The ABS transformer is for auxiliary boiler operation and also provides alternate means to connect the safety related divisional buses to either Line #5 or Line #6 within the plant.

NOTE: Attachment #1 provides a simplified electrical distribution schematic for the 115 KV offsite power supply for Lines #5 and #6.

Plant status prior to events

On the morning of March 23, 1992 prior to the event, the following plant conditions and parameters were in effect:

o NMP2 was at 0 percent power with the reactor mode switch in the "REFUEL" position (Operational Condition 5) and the core partially offloaded (Second Refuel Outage). Fuel handling evolutions were not in progress.

o Division I Emergency Diesel Generator (2EGS*EG1) was inoperable due to planned maintenance.

I. DESCRIPTION OF EVENTS (cont.)

o Division II (2EGS*EG3) and Division III (2EGS*EG2) Emergency Diesel Generators were operable in a standby mode.

o Service Water System (SWP) pumps 2SWP*P1A (Division I) and 2SWP*P1B (Division II) were in operation.

o Normal power to the plant was being supplied as follows: Line #5 aligned to supply Division I, Division III, and normal station service loads through transformer XSR1A with R-50 circuit breaker and motor operated circuit switch MDS-3 closed.

Auxiliary boilers were also being powered from Line #5 with circuit switch MDS-5 and disconnect switch MDS-10 closed. Line #6 was supplying Division if loads and normal station service loads through R-60 circuit breaker and circuit switcher MDS-4.

o Uninterruptible Power Supply 2VBB-UPS1A, which supplies power to Control Room annunciation, was aligned to the maintenance power supply. UPSs are designed such that in the event of loss of normal supply power, alternate auxiliary DC power (batteries) is utilized to power the UPS and the UPS continues to supply the necessary loads. The maintenance supply configuration allows maintenance to be performed on the UPS without interrupting power to the downstream loads, however, auxiliary DC supply power capabilities are bypassed.

o Plant was maintaining a shutdown risk assessment of N + 1 in accordance with NUMARC's shutdown management guidance.

Immediately prior to the event, Relay and Control personnel were assigned to calibrate relay 51-B-2SPRZ18, an overcurrent relay associated with Auxiliary Boiler Service Transformer 2ABS-X1 (relay associated with backup protection for 4160 volt bus 2NNS-SWG018). The design of this particular relay is such that the trip disabling mechanism is under its glass dust cover, therefore requiring removal of the cover to disable the relay. This design means that the relay trips will be active/functional during removal and replacement of the dust cover. During the replacement of the dust cover, at the conclusion of the calibration, the trip reset lever installed in the dust cover inadvertently moved (closed) the contact arm, causing the relay to actuate. The Relay and Control personnel noted that a relay trip had occurred, notified the Control Room, and were directed to the Control Room to review the incident with the SSS.

When the 51-B-2SPRZ18 relay contact madeup, it in turn actuated type 86 lockout relay 86-SPRX11, causing R-50 breaker to open (as designed), thereby tripping Line #5. This in turn de-energized the Division I and Division III Emergency Switchgear and one-half of normal AC station loads. This also resulted in the following:

o Secondary Containment isolated.

I. DESCRIPTION OF EVENTS (cont.)

o Standby Gas Treatment System (GTS) Train B auto started.

o Division III Emergency Diesel Generator (2EGS*EG2) auto started and supplied power to the Division III bus (2ENS*SWG102).

o Loss of the UPS1A functions and the loads connected to it (i.e., Control Room annunciators). UPS alternate power source capabilities were not available (UPS in maintenance supply mode). The loss of annunciation resulted in the declaration of an ALERT emergency classification at NMP2.

o Loss of the Division I Reactor Protection System (RPS) motor generator set.

At 1026 hours, operators attempted to align the de-energized normal AC and Division I emergency AC power distribution to the remaining offsite power source (Line #6) by closing MDS-20.

Neither Control Room operators nor Relay and Control personnel recognized the impact of the tripped Auxiliary Boiler (type 86) relay. Upon closure of MDS-20, control logic for the type 86 relay resulted in opening breaker R-60, thereby tripping Line #6.

Loss of Line #6 resulted in the loss of the remaining offsite power source to the station and the de-energization of all remaining normal AC and the Division II emergency AC power distribution systems. Additionally, this resulted in:

o Loss of shutdown cooling capabilities due to loss of Residual Heat Removal System (RHS) pump 2RHS*P1B.

o Division II Emergency Diesel Generator auto starting and re-energizing the Division II bus.

o Restoration of pump 2RHS*P1B, restoring shutdown cooling capabilities. There was no detectable rise in reactor coolant system temperature.

o Loss of remaining motor generator set, de-energized the scram pilot solenoid valves. De-energizing these solenoids allowed a drain path from the reactor vessel to the scram discharge volume (SDV). The SDV drain and vent valves were open allowing drainage to the Reactor Building equipment drains; however, the SDV filled faster than it could drain, bringing in a high SDV level RPS trip at 1029 hours. This RPS trip closed the SDV drain and vent valves isolating the drain path from the reactor vessel.

I. DESCRIPTION OF EVENTS (cont.)

At 1033 hours, the Division III Emergency Diesel Generator (EDG) tripped on high jacket water temperature due to the isolation of Service Water System flow to the diesel's jacket water coolers.

Operators reset the 115 KV switchyard protective relay and started restoring power via Line #6 at 1044 hours. Power was restored to the "B" reserve station transformer (XSR1B) by 1046 hours and the Auxiliary Boiler Transformer by 1105 hours. Operations personnel then commenced realigning station loads from Line #6; Control Room annunciation was restored at 1131 hours.

At 1152 hours, annunciation was received for a high high level in the Reactor Building floor drain sumps, and operators entered the Emergency Operating Procedures (EOP) for Secondary Containment. The operators restored power to the sumps and commenced pumping the sumps down. Upon clearing the high high level alarm the operators exited the EOPs.

The ALERT was terminated at 1317 hours.

NOTE: Attachment #2 provides a time line for plant conditions during the event.

SUMMARY SERVICE WATER SYSTEM (SWP)

NOTE: Attachment #3 provides a reference diagram of SWP in relation to Divisional Emergency Diesel Generators.

Background

The Service Water System supplies cooling water to the Division III EDG during diesel operations. The cooling water will be supplied by either the Division I or II portions of SWP through the SWP cooling water supply valves 2SWP*MOV95A and B. The position of these valves is controlled by pressure transmitters 2SWP*PT95A and B. These normally open valves receive a signal to close from the SWP supply header pressure transmitters 2SWP*PT95A and B when the header pressure drops below approximately 25 pounds. This setpoint is designed to indicate a line rupture in the supply header downstream of the SWP cooling water supply valves 2SWP*MOV95A and B that could affect the cooling water supply to the Division I and II Diesels. A contact exists in the closing logic of the SWP cooling water supply valves which closes on a time delay, indicating that the Division III Diesel has picked up and has continuously supplied power to the Division III bus for a period of one minute. This time delay is designed to allow SWP time to recover from the decrease in the supply header pressure caused by a loss of offsite power and subsequent trip and restart of the SWP pumps. SWP flow through the Division III Diesel is initiated by the opening of the SWP discharge valves (2SWP*MOV94A/B) after energization of the Division III bus by the diesel. When the Division III Diesel is not operating, the SWP cooling water discharge valves 2SWP*MOV94A and B are normally closed.

I. DESCRIPTION OF EVENTS (cont.)

When offsite power is lost, SWP will trip its running pumps. The SWP supply header pressure will rapidly decrease and the pressure transmitter 2SWP*PT95A and B will trip and send a close signal to the SWP cooling water supply valves 2SWP*MOV95A and B. The Division I and II Diesels will start and pick up their respective buses. The Division III Diesel will start and ten seconds later will pick up the Division III bus 2ENS*SWG102. The one minute time delay relay will start timing and the SWP cooling water discharge valves 2SWP*MOV94A and B will open to provide a cooling water flow path through the Division III Diesel cooler.

Within one minute after the timer starts, the SWP supply header pressure should have recovered and reset the supply header pressure transmitters 2SWP*PT95A and B, thus the SWP cooling water supply valves (2SWP*MOV95A and B) will remain open.

If one minute after the timer starts the SWP supply header pressure has failed to recover and reset, (indicating a line rupture has occurred), then the SWP cooling water supply valves 2SWP*MOV95A and B will close. This action would ensure that the Division I and II Diesels would receive adequate cooling water to operate. Operator action would then be required to reopen these valves.

Event

Prior to the event, cooling water was available to the Division III EDG from Division I and II Service Water pumps 2SWP*P1A and 2SWP*P1B respectively.

At the time of the loss of Line #5, the Division III EDG auto started and supplied power to the Division III bus. Service Water pump 2SWP*P1A tripped with the loss of Division I power, however, water continued to be supplied to the running Division III EDG from Division II pump 2SWP*P1B.

The Division III EDG ran for more than one minute, causing the time delay relay to time out and close its contacts.

With the complete loss of offsite power (loss of Line #6 approximately 18 minutes later), the remaining Service Water pump 2SWP*P1B de-energized and the SWP divisional isolation valve 2SWP*MOV50B closed. SWP header pressure decreased, tripping transmitters 2SWP*PT95A and B. The Division II EDG then successfully auto started, re-energizing the Division II bus and 2SWP*P1B.

This interruption in SWP supply header pressure caused the closure of the Division II SWP supply valve (2SWP*MOV95B) to the running Division III EDG since the one minute duration closure block on the valve had begun 18 minutes earlier with the loss of Line #5. The Division III Emergency Diesel Generator tripped approximately 7 minutes after loss of Line #6 due to high jacket water temperature (loss of cooling water supply).

II. CAUSE OF EVENTS

March 23, 1992 loss of Line #5 at NMP2:

The immediate cause for the loss of Line #5 and the subsequent: (1) auto start of the Division III EDG; (2) Secondary Containment isolation signal; and (3) auto start of the Standby Gas Treatment System Train B was the inadvertent catch of the trip reset lever on the relay flag while Relay and Control personnel were re-installing the glass cover on relay 51-B-2SPRZ18 (located in Relay Room panel 2CEC-PNL802). The actuation of this relay flag accidentally made up the target seal-in contact which in turn operates lockout auxiliary relay 86-2SPRX11 (also located in 2CEC-PNL 802). This in turn actuated a series of other relays which resulted in the tripping of Line #5.

The root cause for this event was determined to be inadequate work practices. Specifically, a failure in the work control process permitted work to be performed on equipment without an adequate pre-work plant impact assessment. This failure permitted authorized Relay and Control personnel to calibrate auxiliary boiler relays which had the potential to trip Line #5, if the technicians performed improperly.

Site Administrative Procedure AP-5.2.5, "Work In Progress (WIP)," requires the inclusion of a WIP data sheet for work being performed on plant equipment. The purpose of the data sheet is to assure work packages receive technical reviews to identify potential equipment/plant impacts and establishing protective barriers prior to commencing work. Upon completion of this review, the data sheet is then processed through the Control Room to provide the Station Shift Supervisor (SSS) and Chief Shift Operator (CSO) assistance in approving work authorization.

The work package associated with the calibration of relay 51-B-2SPRZ18 was developed by Relay and Control, and Work Control Center personnel. Technical reviews were made in accordance with procedures, however, interviews with Control Room supervision, Outage Management, Operations Management, and the technicians performing the work confirmed that consideration had not been given to the risk of tripping Line #5 during the calibration of the relay.

March 23, 1992 loss of Line #6 at NMP2:

The root cause for the loss of offsite power from Line #6 has been determined to be deficiencies in licensed operator training. Control Room operators were unaware of the protective logic associated with the tripped relay from loss of Line #5. Although loss of offsite power is covered in operator training programs, the training focused on recovery actions only, not on protective relaying schemes.

II. CAUSE OF EVENTS (cont.)

Upon completion of extensive interviewing with licensed operators involved in the Line #5 to Line #6 cross tie evaluation, the following behavioral factors were determined to be present during the evolution:

o Unawareness. The operators were not aware of the interlocks associated with the tripped relay, and there were no normal means available to detect the relaying scheme.

o Mindset/Preconceived Idea. The operators approached the task of energizing the switchgear with Line #6 based on the assumption that as long as circuit switcher 2YUC-MDS5 was opened, they were isolated from the signal that tripped Line #5.

o Wrong Assumptions Made. The supposition was made that the tripping signal was on the load side of the Auxiliary Boiler Transformer. This assumption allowed the relay to remain tripped during the cross tie evolution.

o Lack of Specific System or Component Knowledge. Training has not been provided on the relaying scheme of the tripped device.

This behavioral factor is closely related to the "Unawareness" behavioral factor discussed above.

o Perceived pressure to complete task. The assessment of evolving plant conditions and parameters following the loss of Line #5 directed operators to react to the urgency of the situation. The perceived pressure hindered operators questioning attitude toward the tasks at hand and precluded them from discovering the interlock.

Contributing factors to this root cause were:

Verbal communication - inadequate communication between Operations and Relay and Control personnel resulted in failure to transmit pertinent information concerning the line cross tie evolution. Transmittal of this information could have prevented this event.

Written communication - The Operating Procedure for the Line #5/Line #6 cross tie evolution (N2-OP-70, "Station Electrical Feed and 115 KV Switchyard"), omitted relevant information.

Tripping of Division III Emergency Diesel Generator (2EGS*EG2) on high jacket water temperature following a sequential Loss of Offsite Power (LOOP):

The cause of the HPCS Diesel Generator trip upon closure of the inlet cooling water valves was the occurrence of an event that the plant was not required to be analyzed for in its licensing bases. Specifically, the sequential Loss of Offsite Power (LOOP), half LOOP (loss of Line #5), followed by a delayed half LOOP (loss of Line #6), was an action not considered in the design

bases for NMP2. The plant design bases does consider a full LOOP, and had a full loss of power occurred during the March 23, 1992 incident, the HPCS Diesel would have continued to operate satisfactory.

During the review of this incident, Niagara Mohawk determined two additional scenarios within the design bases could be postulated where the HPCS Diesel Generator would lose its cooling water. The two new scenarios would have the HPCS Diesel Generator running for a test or as a result of a low reactor water level initiation with either followed by a delayed LOOP. In the previous design configuration, the cooling water to the HPCS Diesel would isolate in either scenario.

The root cause for these two identified conditions was a design deficiency. With the HPCS Diesel Generator running in the test mode with a delayed LOOP, the loss of service water would result in a low pressure isolation of the cooling water to the HPCS Diesel Generator. This would render the HPCS Diesel Generator unavailable for accident mitigation due to a high temperature trip without operator action to restore service water from the Control Room.

Should the HPCS Diesel Generator be running in response to a LOCA signal and the plant experience a delayed LOOP, the loss of service water would result in a low pressure isolation of the cooling water to the HPCS Diesel Generator. During a LOCA, the high temperature trip of the HPCS Diesel Generator is bypassed. As a result, high temperature damage will occur to the HPCS Diesel Generator without operator action to trip the HPCS Diesel Generator or restore cooling water. This would render the HPCS Diesel Generator unavailable for accident mitigation due to high temperature damage.

The design change which has been implemented during the plant's Second Refueling Outage to prevent loss of the Diesel Generator during the scenarios discussed above will also preclude the closure of the valves under the sequence of events experienced during the March 23, 1992 event.

III. ANALYSIS OF EVENTS

The following conditions are reportable in accordance with 10CFR50.73 (a)(2)(iv), "any event or condition that results in manual or automatic actuation of an Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

o Secondary Containment isolation
o Initiation of Standby Gas Treatment System Train B

III. ANALYSIS OF EVENTS (cont.)

o Reactor Scram
o Automatic Start of Division II and III Emergency Diesel Generators

The following condition is reportable in accordance with 10CFR50.73(a)(2)(ii)(B), "any event or condition that resulted in the condition of the nuclear power plant being in a condition that was outside the design basis of the plant."

o Design deficiency in the High Pressure Core Spray (HPCS) Diesel Generator (Division III) jacket cooling water supply valve isolation logic which could have rendered the diesel inoperable.

Secondary Containment isolation/initiation of Standby Gas Treatment (GTS)

The Standby Gas Treatment System is an Engineered Safety Feature (ESF) designed to limit the release of radioactive gases from the Reactor Building to the environment within the guidelines of 10CFR100 in the event of a Loss of Coolant Accident (LOCA). GTS can also be used to maintain normal Reactor Building differential pressure when the normal Reactor Building Ventilation System is not available. A Secondary Containment isolation and the initiation of GTS are considered conservative ESF plant responses with minimal plant impact and no resultant impact on public safety.

Reactor scram

The scram discharge volume consists of header piping that connects to each hydraulic control unit (HCU) and drains into an instrument volume. The header piping is sized to receive and contain all the water discharged by the drives during a scram, independent of the instrument volume.

During normal plant operation the scram discharge volume is empty and vented to atmosphere through its open vent and drain valves. When a reactor scram occurs, upon a signal from the safety circuit, these vent and drain valves are closed to conserve reactor water. Lights in the main Control Room indicate the position of these valves. Redundant vent and drain valves assure against loss of reactor coolant from the SDV following a scram.

The loss of Line #6 resulted in the loss of the remaining motor generator set for the Reactor Protection System (RPS), which de-energized the scram pilot solenoid valves. De-energizing these solenoids allowed a drain path from the reactor vessel to the scram discharge volume (SDV). The SDV drain and vent valves (which are normally open) were open allowing drainage from the SDV to the Reactor Building equipment drain tanks.

It can be determined from the plant data that the drain path was established approximately one

expected to have kept the scram pilot solenoids energized for approximately 2 minutes after the loss of Line #6. After the scram pilot solenoid valves de-energized, the SDV filled faster than it could drain, actuating an RPS trip signal (on high SDV water level) approximately three minutes after losing Scriba Line #6. This RPS signal closed the

SDV vent and drain valves isolating the drain path.

If the plant had been at rated power, the SDV would accumulate water at a rate faster than the March 23rd event. Therefore, the drain path would exist for a shorter time period. Additionally, operators would have inserted a manual scram had this event occurred during power operations when an automatic scram could not be verified. This action also would have isolated the drain path.

This RPS trip signal is a conservative plant response which does not pose any safety consequences.

Automatic start of the Division II and Division III Emergency Diesel Generators

The Standby Diesel Generators are designed to provide onsite power to the loads necessary to bring the plant to a safe shutdown condition following a LOCA and loss of offsite power. They also provide power to bring the plant to a safe shutdown condition after an extended loss of offsite power.

Immediately prior to the loss of Line #5, the Division II and III Standby Diesel Generators, 2EGS*EG3 and 2EGS*EG2 respectively, were in standby to support associated buses in the event of loss of offsite power. With the loss of Lines #5 and #6, the Division II and III Diesels started and restored power to the respective buses as designed.

March 23, 1992 event at full Dower coincident with a LOCA

An evaluation of the incident concluded that the Division II and III EDGs functioned as designed. it was further concluded that during normal power operation, the Division II and III EDGs would have responded as described in the NMP2 USAR to a full LOOP coincident with a LOCA. However, the high temperature trip of the Division III EDG following a partial LOOP (loss of Scriba line #5) followed by a complete LOOP (loss of Scriba line #6), is not specifically analyzed in the USAR from an ECCS/large break LOCA perspective.

Table 6.3-3 of the USAR details the single active failures, including a Division I EDG failure, considered in the ECCS performance evaluation. Since the Unit's limiting conditions of operation allow an EDG to be out of service for a maximum of only 72 hours, we can evaluate this situation considering the normal availability of all three EDGs. The March 23, 1992 event with the Division I EDG out of service for maintenance is analogous to an active failure of that EDG. The NMP2 ECCS performance evaluation concluded that the ECCS could maintain

III. ANALYSIS OF EVENT (cont.)

adequate core cooling in the event of a LOCA, coincident with a LOOP and a failure of the Division I EDG.

The USAR design basis for the Reactor Core isolation Cooling (RCIC) System explains that if High Pressure Core Spray (HPCS) fails and the RCIC capacity is insufficient, the Automatic Depressurization System (ADS) will automatically depressurize the RPV to permit the low pressure ECCS to provide makeup coolant. But the USAR does not specifically evaluate whether the Division II ECCS systems alone (2 low pressure ECCS pumps and ADS), in conjunction with RCIC, could provide adequate core cooling in the unlikely event of a large break LOCA, coincident with a LOOP and failures of the Division I and III Emergency Diesel Generators.

NMP2 USAR Table 6.3-3, based upon General Electric's (GE) GESTAR II accident methodology calculations, represents evaluations of large-break LOCAs and ECCS LOCAs, including a break in the HPCS. Section 6.3.1.1.2 of the USAR evaluates HPCS disabled by a line break and a failure of the Division I EDG, coincident with a LOOP. This section concludes that two LPCI loops and ADS would be available to provide adequate core cooling for an ECCS LOCA.

For large-break LOCAs, a recent analysis performed by GE using the SAFER/GESTR accident methodology reached a similar conclusion. The SAFER/GESTR analysis, assuming only two (2) Division II Low Pressure Coolant Injection System (LPCI) Loops available, resulted in peak clad temperature of approximately 1500 degrees Fahrenheit, well below the 2200 degrees Fahrenheit limit of 10CFR50.46.

It is therefore concluded that, had the March 23, 1992 sequence of events occurred at full power operation, following a LOCA, the plant could have attained cold shutdown while maintaining adequate core cooling.

IV. CORRECTIVE ACTIONS

Immediate actions taken following the trip of Line #5 were: to verify fuel moves were not in progress, establish communication with the fuel floor to continuously monitor Spent Fuel Pool and Reactor Cavity water levels; and assess action required to cross tie Line #6.

Immediate actions taken following loss of Line #6 included: verification of auto start of Division II EDG; restart of 2RHS*P1B (restoration of RHS

Shutdown Cooling Mode); verification that reactor coolant temperature remained consistent at 95 degrees Fahrenheit.

IV. CORRECTIVE ACTIONS (cont.)

Additional corrective actions taken:

1. Site direction was given by Plant Managers (Units 1 and 2) detailing expectations for pre-work job planning of Relay and Control Work In Progress data sheets as outlined in Site Administrative Procedure AP-5.2.5, "Work in Progress (WIP)."

2. An assessment organization was developed by the Plant Manager to investigate the circumstances leading to this event.

3. A training mockup has been developed to certify Relay and Control personnel for removal/ re-installation of relay covers.

4. The Relay and Control Department has enhanced pre-job briefings to ensure personnel are familiar with plant/equipment impact during testing activities.

5. A training effectiveness evaluation was conducted by the Operations Training Department to review and identify operator strengths and weaknesses resulting from this event. This review resulted in the modification of existing, and the development of new Requalification Lesson Plans to upgrade licensed and non-licensed operator training programs. The plans include but are not limited to:

o The review of the Auxiliary Boiler Relay protection logic and how it affects offsite power availability.

o Plant impacts upon loss of individual UPS loads.

o A technical scenario by the Relay and Control Department on line protection schemes and relaying.

Additionally, simulator evaluations identified that a reproduction of a similar loss of Line #5 scenario was not possible. A simulated discrepancy report was generated to correct deficiency.

6. Operations management has discussed the miscommunication aspects of the March 23, 1992 event with Operations personnel, and developed guidelines for a more assertive and inquisitive type of

communication between Operations and Support personnel.

7. Operating Procedure N2-OP-70, "Station Electrical Feed and 115 KV Switchyard," has been revised to include protective device relaying schemes to aid operators in correcting alarm conditions.

IV. CORRECTIVE ACTIONS (cont.)

8. Engineering Department has completed an evaluation of interlocks/switch logic for breaker/disconnect switches associated with Lines #5 and #6. Review results determined that all schemes are consistent with NMP2 design philosophy.

9. A plant modification (PN2Y92MX006) has been completed on the Service Water System supply valve logic for the Division III Emergency Diesel Generator. This change included:

A. Installing a new time delay in the logic from the SWP supply header pressure transmitters 2SWP*PT95A and B to SWP supply valves 2SWP*MOV95A and B to allow a minimum of 70 and a maximum of 96 seconds prior to closing these valves due to low SWP supply header pressure.

B. Removing the existing time delay in the logic from the diesel to the SWP supply valves 2SWP*MOV95A and B to prevent the loss of the diesel due to the closing of the valves after the diesel has been running for greater than 1 minute and on any low pressure signal from the pressure transmitters thereafter.

V. ADDITIONAL INFORMATION

A. Failed components: none.

B. Previous similar events:

NMP2 has experienced and reported loss of one of the two offsite power feed lines (LER 91 -12). However, the plant has not experienced an event involving the loss of both offsite power sources.

V. ADDITIONAL INFORMATION (cont.)

C. Identification of components referred to in this LER:

COMPONENT IEEE 803 IEEE 805
EIIS FUNCTION SYSTEM ID

Reactor Building Ventilation System N/A VA
Standby Gas Treatment System N/A BH
Emergency Diesel Generator System N/A EK
Service Waster System N/A BI
High Pressure Core Spray System N/A BG
Reactor Protection System N/A JC
Switchyard N/A FK
Residual Heat Removal System N/A BRO
Emergency Distribution N/A BAYOU
Secondary Containment N/A NG
Battery BTRY BAYOU
Control Room Annunciation N/A IB
Main Transformer XFMR EL
Uninterruptible Power Supply UJX EE
Bus BU EK
Diesel Generator DG EK
Pump P BI, BRO
Number 5 and 6 Feeder Line FDR FK

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ATTACHMENT NO. 1

Figure "Simplified Electrical Distribution" omitted.

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ATTACHMENT 2

EVENTS

TIME (3/26/92) ACTIVITY

10:08:02 o Loss of Line #5 (115 KV).
o Loss of Uninterruptible Power Supply 2VBB-UPS1A
loads.
o Loss of Control Room annunciation.

o Secondary Containment isolation signal.
o Standby Gas Treatment System Train B autostart.
o Division III Emergency Diesel Generator

(2EGS*EG2) auto starts.
o Service Water System pump 2SWP*P1A (Div. 1)
trips.

10:16 o Station Shift Supervisor (SSS) declares 'ALERT'
Emergency.

10:26:57 o Loss of Line #6 (115 KV).
o Division II Emergency Diesel Generator
(2EGS*EG3) auto starts.
o Loss of Residual Heat Removal pump 2RHS*P1B
(loss of shutdown cooling).

10:28:27 o 2RHS*P1B restored (reactor coolant temperature
prior to and after restoration of shutdown
cooling was constant 95 degrees Fahrenheit).

10:29:36 o RPS Logic full scram due to scram discharge
volume high level.

10:33:50 o Division III Emergency Diesel Generator trips
off line (high jacket H sub 20 temperature).

11:44 o Power restored through Line #6 (via Auxiliary
Boiler Transformer).

11:53 o Secured GTS Train 'B'.

12:27 o Scram reset.

12:45 o Division II Emergency Diesel Generator off grid.

13:17 o 'ALERT' emergency terminated.

ATTACHMENT NO. 3

Figure "SERVICE WATER SYSTEM SUPPLY" omitted.

ATTACHMENT 1 TO 9206250028 PAGE 1 OF 1

NM NIAGARA
MOHAWK
NINE MILE POINT-UNIT 2/P.O. BOX 63, LYCOMING, NY 13093

Martin J. McCormick Jr. PE NMP87217
Plant Manager-Unit 2
Nuclear Generation

June 15, 1992

United States Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

RE: Docket No. 50-410
LER 92-06, Supplement 1

Gentlemen:

In accordance with 10CFR50.73, we hereby submit the following Licensee Event Report:

LER 92-06 Supplement 1 is being submitted in accordance with the following:

1. 10CFR50.73 (a)(2)(iv), "any event or condition that results in manual or automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS)."

2. 10CFR50.73 (a)(2)(ii)(B), "any event or condition that resulted in the condition of the nuclear power plant being in a condition that was outside the design basis of the plant."

A 10CFR50.72 (b)(2)(ii) report was made at 1035 hours on March 23, 1992.

A 10CFR50.72 (b)(2)(iii)(D) report was made at 1717 hours on April 17, 1992.

This Supplement is being issued to provide the results of the investigation into the design conditions in the High Pressure Core Spray (HPCS) Diesel Generator cooling water supply valve isolation logic that led to loss of the HPCS Diesel during this event.

This report was completed in the format designated in NUREG-1022, Supplement 2, dated September 1985.

Very truly yours,

Martin J. McCormick Jr.
Plant Manager - NMP2

MJM/GB/lmc
ATTACHMENT

xc: Thomas T. Martin, Regional Administrator Region I
Wayne L. Schmidt, Senior Resident Inspector

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